NUREG-0800
(Formerly NUREG-75/087)

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

15.2.6 LOSS OF NONEMERGENCY AC POWER TO THE STATION AUXILIARIES

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - None

I. AREAS OF REVIEW

The loss of nonemergency ac power is assumed to result in the loss of all power to the station auxiliaries. This situation could result either from a complete loss of the external grid (offsite) or a loss of the onsite ac distribution system. It is different from the loss of load condition considered in Standard Review Plan (SRP) Section 15.2.2 because, in the latter case, ac power remains available to operate the station auxiliaries. The major difference is that in the loss of ac power transient all the reactor coolant circulation pumps are simultaneously tripped by the initiating event. This causes a flow coastdown as well as a decrease in heat removal by the secondary system.

Within a few seconds the turbine trips and the reactor coolant system is isolated, causing the pressure and temperature of the coolant to increase. A reactor trip is initiated. The diesel generators are automatically started and provide electric power to the vital loads. The sensible and decay heat loads are handled by actuation of the steam relief system, steam bypass to the condenser, reactor core isolation cooling system in a boiling water reactor (BWR), emergency core cooling system, (BWR) and auxiliary feedwater system in a pressurized water reactor (PWR).

The review of the loss of ac power transient includes the sequence of events, the analytical model, the values of parameters used in the analytical model, and the predicted consequences of the transient.

The sequence of events described in the applicant's safety analysis report (SAR) is reviewed by both RSB and the Instrumentation and Control Systems Branch (ICSB). The RSB reviewer concentrates on the need for the reactor protection system, the engineered safety systems, and operator action to secure and maintain the reactor in a safe condition.

The analytical methods are reviewed by RSB to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the reviewer requests initiation of a generic evaluation of the new analytical model by RSB or the Core Performance Branch (CPB) as appropriate.

The predicted results of the transient analysis are reviewed to assure that the consequences meet the acceptance criteria given in subsection II, below. The results of the analysis are reviewed to ascertain that the values of pertinent system parameters are within expected ranges for the type and class of reactor under review.

The RSB will coordinate other branch evaluations that interface with the overall review of the transient analysis as follows: The ICSB reviews the instrumentation and controls aspects of the sequence described in the SAR to evaluate whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems. The ICSB evaluates the design of the auxiliary feedwater system to determine that the requirements and guidance of II.E.1.2 of NUREG-0737 are met. The RSB reviewer consults with the ICSB reviewer to assure that the appropriate delay time for auxiliary feedwater initiation is assumed in the analysis.

The reliability of the auxiliary feedwater system is reviewed by the ASB in accordance with SRP Section 10.4 and in accordance with the requirements and guidance of II.E.1.1 of NUREG-0737 and II.K.2.(1) (item 1 of Table C.2) of NUREG-0660. The RSB reviewer consults with the ASB reviewer to assure that the operational assumptions for the auxiliary feedwater system in the analysis is appropriate. As part of its primary review responsibility for SRP Sections 7.2 through 7.5, the Core Performance Branch (CPB), upon request from RSB, reviews the values of the parameters used in the analytical models which relate to the reactor core for conformance to plant design and specified operating conditions; determines the acceptance criteria for fuel cladding damage limits; and reviews the core physics, fuel design, and core thermal-hydraulics data used in the SAR analysis as part of its primary review responsibility for SRP Sections 4.2 through 4.4. The Accident Evaluation Branch (AEB), using fuel damage results provided by RSB, evaluates the radiological consequences associated with fuel failure. The review of the technical specifications is coordinated and performed by the Licensing Guidance Branch (LGB) as part of its primary review responsibility for SRP Section 16.0.

For those areas of review identified above as being reviewed as part of the primary review responsibility of other branches, the acceptance criteria necessary for the review and their methods of application are contained in the referenced SRP section of the corresponding review branch.

II. <u>ACCEPTANCE CRITERIA</u>

The RSB acceptance criteria are based on meeting the relevant requirements of the following regulations:

A. General Design Criterion 10 as it relates to the reactor coolant system being designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during normal operations including anticipated operational occurrences.

B. General Design Criterion 15 as it relates to the reactor coolant system and its associated auxiliaries being designed with appropriate margin to assure that the pressure boundary will not be breeched during normal operations including anticipated operational occurrences.

C. General Design Criterion 26 as it relates to the reliable control of reactivity changes to assure that specified acceptable fuel design limits are not exceeded, including anticipated operational occurrences. This is accomplished by assuring that appropriate margin for malfunctions, such as stuck rods, are accounted for.

D. TMI Action Plan items II.E.1.1, II.E.1.2, and II.K.2(1) of NUREGs-0718 and -0737 as they relate to the performance requirements of the auxiliary feedwater system for the loss of nonemergency ac power event.

Specific criteria necessary to meet the relevant requirements of GDC 10, 15, and 16 for events of moderate frequency* are as follows:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values (Ref. 1).

2. Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit for PWRs and the CPR remains above the MCPR safety limit for BWRs based on acceptable correlations (see SRP Section 4.4).

3. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

4. An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered and is an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. For such accidents, fuel failures must be assumed for all rods for which the DNBR or CPR falls below those values cited above for cladding integrity unless it can be shown, based on an acceptable fuel damage model (see SRP Section 4.2), that fewer failures occur. There shall be no loss of function of any fission product barrier other than the fuel cladding.

5. To meet the requirements of General Design Criteria 10 and 15, the positions of Regulatory Guide 1.105, "Instrument Spans and Setpoints," are used with regard to their impact on the plant response to the type of transient addressed in this SRP section.

6. The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR Part 50, shall be identified and assumed in the analysis and shall satisfy the positions of Regulatory Guide 1.53 (Ref. 14).

The applicant's analysis of the loss of ac power transient should be based on an acceptable model. Models which have been approved by the NRC are identified in References 2 through 8. If the applicant proposes analytical methods which have

*The term "moderate frequency" is used in this SRP section in the same sense as in the definitions of design and plant process conditions in References 9 and 10.

not been approved, these are evaluated by the staff for acceptability. For new generic methods, the reviewer requests an evaluation by the appropriate branch.

The value of parameters used in the analytical model should be suitably conservative. The following values are considered acceptable for use in the model.

a. The initial power level is taken as the licensed core thermal power for the number of loops initially assumed to be operating plus an allowance of 2% to account for power measurement uncertainties, unless a lower power level can be justified by the applicant. The number of loops operating at the initiation of the event should correspond to the operating condition which maximizes the consequences of the event.

b. Conservative scram characteristics are assumed, i.e., for a PWR - maximum time delay with the most reactive rod held out of the core, and for a BWR - a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate.

c. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, power profile and radial power distribution.

d. Mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument inaccuracy in accordance wih Regulatory Guide 1.105. Compliance with Regulatory Guide 1.105 is determined by ICSB.

III. REVIEW PROCEDURES

The procedures below are used during the review of both construction permit (CP) and operating license (OL) applications. During the CP review the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review stage, final values should be used in the analysis and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

The description of the loss of ac power transient presented by the applicant in the SAR is reviewed by RSB regarding the occurrences leading to the initiating event. The sequence of events from initiation until a stabilized condition is reached is reviewed to ascertain:

1. The extent to which normally operating plant instrumentation and controls are assumed to function.

2. The extent to which plant and reactor protection systems are required to function.

3. The credit taken for the functioning of normally operating plant systems.

4. The operation of engineered safety systems that is required.

5. The extent to which operator actions are required.

6. The operation of standby diesel generators that is required.

7. That appropriate margin for malfunctions, such as stuck rods (per II.3.b above) are accounted for.

If the SAR states that the loss of ac power transient is not as limiting as some other similar transient, the reviewer evaluates the justification presented by the applicant. If a quantitative analysis of the loss of ac power transient is presented in the SAR, the RSB reviewer, with the aid of the ICSB reviewer, reviews the timing of the initiation of those protection, engineered safety, standby diesel generator, and other systems needed to limit the consequences of the transient to an acceptable level. The RSB reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints. The ICSB review of Chapter 7 of the SAR confirms that the instrumentation and control systems design is consistent with the requirements for safety systems actions for these events.

To the extent deemed necessary, the RSB reviewer evaluates the effects of single active failures of systems and components which may affect the course of the transient. This aspect of the review uses the procedures described in SRP sections for Chapters 4, 5, 6, 7, 8, and 9 of the SAR.

The mathematical models used by the applicant to evaluate core performance and to predict system pressure in the reactor coolant system and main steam lines are reviewed by RSB to determine if these models have been previously reviewed and found acceptable by the staff. If not, a generic review of the model proposed by the applicant is initiated.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by RSB. Of particular importance are the reactivity coefficients and control rod worths used in the applicant's analysis, and the variation of moderator temperature, void, and Doppler coefficients of reactivity with core life. The justification provided by the applicant to show that he has selected the core burnup that yields the minimum margins is evaluated. CPB is consulted regarding the values of the reactivity parameters used in the applicant's analysis.

The results of the analysis are reviewed and compared to the acceptance criteria presented in subsection II of this SRP section regarding the maximum pressure in the reactor coolant and main steam systems. The variations with time during the transient of neutron power, heat fluxes (average and maximum), reactor coolant system pressure, minimum DNBR (PWR) or CPR (BWR); core and recirculation loop coolant flow rates (BWR), coolant conditions (inlet temperature, core average temperature (PWR), core average steam volume fraction (BWR), average exit and hot channel exit temperatures, and steam fractions), steam line pressure, containment pressure, pressure relief valve flow rate, and flow rate from the reactor coolant system to the containment system (if applicable) are reviewed. The more important of these parameters for the loss of ac power transient are compared to those predicted for other similar plants to verify that they are within the expected range.

IV. EVALUATION FINDINGS

The evaluation findings under this SRP section are incorporated in a statement covering all transients of moderate frequency involving a decrease in heat removal by the secondary system. See the findings statement in SRP Section 15.2.1-5 for a typical statement.

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guides and NUREGs.

VI. REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

2. NUREG-0151, Safety Evaluation Report, "GESSAR-251, Nuclear Steam Supply System Standard Design," General Electric Company, March 1977.

3. NUREG-0152, Safety Evaluation Report, "GESSAR-238, Nuclear Steam Supply System Standard Design," General Electric Company, March 1977.

4. NUREG-75/103, Safety Evaluation Report, "RESAR-41 Standard Reference System," Westinghouse Electric Corporation, December 1975.

5. NUREG-0104, Safety Evaluation Report, "RESAR-35, Standard Reference System," Westinghouse Electric Corporation, December 1976.

6. NUREG-0491, Safety Evaluation Report, "RESAR-414 Standard Reference System," Westinghouse Electric Corporation, November 1978.

7. NUREG-75/112, Safety Evaluation Report, "CESSAR System 80, Standard Reference System," Combustion Engineering Incorporated, December 1975.

8. NUREG-0433, Safety Evaluation Report, "B-SAR-205, Nuclear Steam Supply System," Babcock & Wilcox Company, May 1978.

9. ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," American National Standards Institute (1974).

10. ANS Trial Use Standard N212, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants," American Nuclear Society (1974).

11. General Design Criterion 10, "Reactor Design."

12. General Design Criterion 15, "Reactor Coolant System Design."

13. General Design Criterion 21, "Protection System Reliability and Testability."

14. Regulatory Guide 1.53, "Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems."

15. Regulatory Guide 1.105, "Instrument Spans and Setpoints."

16. NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing Licenses."

17. NUREG-0737, "Clarification of TMI Action Plan Requirements."

18. General Design Criterion 26, "Reactivity Control System Redundancy and Capability."